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TITLE: RECENT JOINT DEVELOPMENTS IN CROSS-SECTION UNCERTAINTY ANALYSIS AT LOS ALAMOS AND EIR

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Recent Joint Developments in Cross-Section Uncertainty Analysis at Los Alamos and EIR

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1 Introduction

Since the last IAEA meeting on this topic, the pace of sensitivity and uncertainty analysis studies has slackened relative to those of the 1970s. The paucity of published studies, at least in the U.S.A., has not reflected a diminished interest in or importance of uncertainty analysis. Rather, it is a natural result of the indefinite postponement of plans for a next-generation fusion device burning DT and for a fusion materials irradiation test facility. Without a specific project as a successor to the TFTR, nuclear systems have been of secondary interest in the fusion program. However, recent emphasis on developing an integral experiment capability in support of fusion reactor blanket/shield analysis has led to renewed uncertainty analysis requirements. As for previous fusion reactor studies[1,2,3], a two-dimensional computational capability is required for improved accuracy of the analysis. This same capability will also be required for emerging fusion reactor design studies (e.g., the ETF in the U.S.A. and the NET in Western Europe).

The review of sensitivity and uncertainty methods, codes, and applications presented by one of the authors[4] at the 1978 IAEA meeting will not be updated here. A subsequent review[5] of the status of nuclear data (including covariances), sensitivity and uncertainty methods, and transport methods and codes is still reasonably timely in most respects. In the present paper we will restrict the discussion to recent developments and future plans for the SENSIBL code (the successor to the SENSIT[6] and SENSIT-2D[7] codes), along with associated covariance data and cross-section libraries.

While the original impetus to SENSIT-2D development was the Fusion Engineering Device (FED) project, renewed interest in developing and applying the code has come from ongoing fusion nucleonic integral experiments in Japan and Switzerland. Experiments at the Fusion

Neutron Source (FNS) facility at JAERI are being analyzed in a cooperative U.S./Japanese effort. A second major program of fusion nucleonics integral experiments is being conducted at the LOTUS facility in Lausanne, Switzerland. There the Lithium Breeding Module (LBM) constructed for the U.S. Electric Power Research Institute is being used by the Swiss Federal Polytechnic School (EPFL) for a series of tritium breeding experiments. An active analysis effort at EPFL, EIR and Los Alamos is underway to compare experimental data with computations using state-of-the-art nucleonic codes and cross-section data. In particular, an intensive joint effort by Los Alamos and EIR is being pursued in uncertainty analysis of the calculated tritium breeding data. These joint efforts are under the umbrella of an agreement of cooperation in fusion reactor nucleonics between EIR and Los Alamos[8,9]. Since 1982 several joint efforts under the agreement have been undertaken, including continued development of cross-section processing (the NJOY code[10]), sensitivity and uncertainty methods (the SENSIBL code), and transport methods (the TRISM code[11]). Some of the Los Alamos effort, especially the development of the COVFILS-2 multigroup covariance library[12], has also been in support of the U.S./Japan cooperation concerning integral experiments at the FNS. Perhaps it is of interest to note in passing that the development of covariance libraries and a two-dimensional sensitivity and uncertainty analysis code is responsive to recommendations of the IAEA Working Group on Neutron Transport and Gamma-ray Production[13].

Briefly, since the 1978 IAEA meeting there has been significant progress in providing both covariance data and multidimensional sensitivity and uncertainty analysis codes. Covariance data are much more prevalent in ENDF/B-V than in earlier evaluated data files, and several multigroup covariance libraries have been produced. Multidimensional sensitivity calculations have been performed by several researchers[2,3,14], using both multigroup deterministic and Monte Carlo transport methods. However, these data and codes are still under development and only now are extensive applications to analyses of integral experiment being undertaken. Perhaps by the next meeting in this series we will have a reservoir of experience and hence intuition regarding the uncertainties in fusion reactor design parameters caused by nuclear data uncertainties.

2 Calculational Methods

The calculational methods used for fusion blanket analysis at EIR and Los Alamos are basically identical and are shown in Figure 1. TRISM is a computer program for solving the two-dimensional neutral particle transport equation in rectangular (X,Y) and cylindrical (R,Z) geometries within a general domain having curved or other nonorthogonal boundaries. The spatial discretization is accomplished using triangular finite elements and discontinuous linear trial functions. TRISM is a follow-on version of TRIDENT-CTR[15] that includes deterministic streaming capabilities[16]. The use of this deterministic streaming option is useful in mitigating the inaccuracies due to the "ray effect" which plague calculations for fusion systems with large internal void regions. The use of triangles in R,Z geometry allows a user to accurately follow curved or irregularly shaped boundaries and material interfaces of toroidal and other fusion

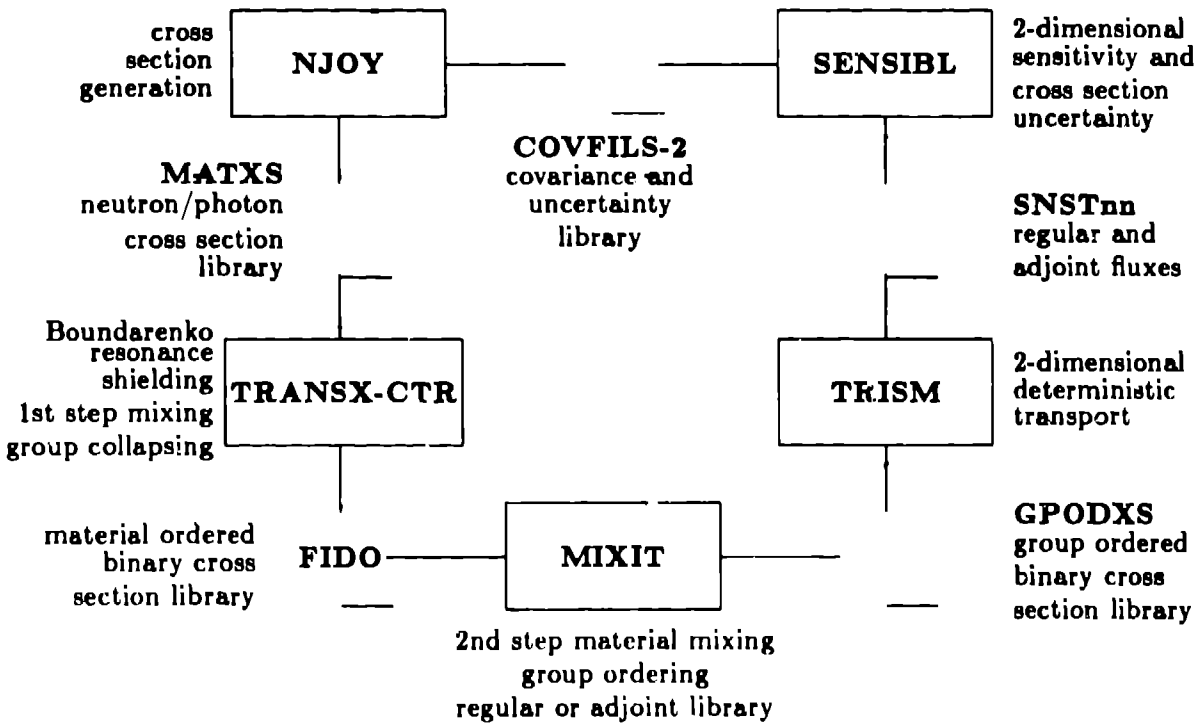

Figure 1: Calculational Scheme

system shapes. TRISM maintains all of the capabilities of TRIDENT-CTR but incorporates a completely new user-friendly free-field input format similar to that of ONEDANT[17] and TWODANT[18]. In addition, several new input and edit options have been added. MIXIT[19] is the code for second-step homogenizations and group-ordered library production.

SENSIBL is an improved and accelerated version of SENSIT-2D, which was an extension of the one-dimensional sensitivity code SENSIT[6]. SENSIBL has the capability for cross-section sensitivity and uncertainty analysis, secondary-energy-distribution (SED) sensitivity and uncertainty analysis, and design sensitivity analysis. The algorithms used are based on first-order generalized perturbation theory. The code allows X,Y or R,Z geometry options and accepts group-dependent quadrature sets. It is intended for use with the two-dimensional, multigroup, discrete-ordinates transport code TRISM. The triangular mesh used by TRISM allows unique modelling capabilities which are applicable to fusion reactor configurations, and thus SENSIBL can also analyze these configurations. The forward and adjoint angular fluxes generated by TRISM are required input to SENSIBL. Because the number of angular fluxes can be voluminous, a sophisticated data management scheme was necessary for the code to keep the execution

time and memory requirements within reasonable limits. As discussed in the following sections, SENSIBL incorporates a number of recently implemented improvements in SENSIT-2D, intended both to maintain consistency with COVFILS-2 and to add new calculational capabilities.

3 Cross-Section Data Libraries

The general-purpose MATXS8 coupled 187-neutron/24-photon group library, based on ENDF/B-V evaluations, is used as the basic library for transport calculations at Los Alamos. This multigroup library contains 31 isotopes and was produced in October 1983 using the NJOY system. It also contains Los Alamos evaluations for ^{7}Li, ^{182}W, ^{183}W, ^{184}W, and ^{186}W, as well as the ENDF/B-V.2 version of Fe(nat). The temperature is 300 K and a thermal + 1/E + fission + fusion weighting spectrum is used. For all isotopes heating data (kerma), and for most important isotopes radiation-damage-energy production data, are available.

In the framework of the common Los Alamos/EIR analysis of the LBM experiments at the LOTUS facility[20,21,22], a new multigroup library was constructed at EIR from JEF-1 and the European Fusion File (EFF) using the same 187-neutron group structure. For photon production and interaction cross sections, the Los Alamos 48-group structure was selected. The pointwise neutron and photon files (PENDF) based on JEF-1 evaluations were produced using the June 1983 version of the NJOY system[10,23,24]. This neutron library was generated by obtaining ^{9}Be and ^{7}Li from a preliminary version of EFF developed presently under the leadership of Euratom. Those basic pointwise neutron files were reconstructed for temperatures from 296 to 3000 K, but only 296 K is included in the present groupwise library.

The multigroup library includes vectors for all reaction types, matrices for reactions producing neutrons (including fission), and data pertaining to fission yields of prompt and delayed neutrons. Furthermore, different kinds of gamma-ray production matrices, partial cross sections, as well as heating and damage data, were also processed. The CLAW weighting spectrum (cf. Ref. [24]) was used and a P_5 order of scattering was included. The most important resonances were shielded using the Bondarenko method. The library presently contains the 83 isotopes listed in Ref. [19].

COVFILS-2[12,27] is a library of multigroup neutron cross sections, scattering matrices, and covariances (uncertainties and their correlations). The 14 materials included in the first version of COVFILS-2 are ^{1}H, ^{6}Li, ^{7}Li, ^{9}Be, C(nat), ^{14}N, ^{16}O, ^{23}Na, ^{27}Al, Si(nat), Cr(nat), Fe(nat), Ni(nat), and Pb(nat). COVFILS-2 was produced using various modules of the NJOY nuclear data processing system[10,25]. It is largely based on data evaluations from ENDF/B-V, although some minor corrections and improvements are incorporated. In cases where the covariance evaluation is missing (as in the case of Be) or judged to be inadequate, private Los Alamos evaluations[26] are employed. The 74-group structure[12] was chosen for compatibility with the 187-group MATXS8.

4 Recent Developments

The theoretical foundations of cross-section sensitivity and uncertainty analysis are well documented in the literature (cf. Ref. [4] and other works cited there). For reference during the discussions below, we list here the principal expressions resulting from classical perturbation theory. For our subsequent development, we view the relative covariances (in COVFILS-2, for example) as *microscopic* cross-section covariances, handling possible spatial variations of the nuclide density within the calculation of the sensitivity of responses to changes in these microscopic data.

Given a set of multigroup microscopic cross-section data, σ^g_{xm}, we are interested in an expression for the standard deviation ΔI of a response I. Examples for I include kerma, displacements per atom (dpa), activation rates, or tritium breeding. The definition of I includes a specification of the spatial region over which the response is to be integrated (the "detector" region). Using the concepts of sensitivity profiles and covariance data, one has a straightforward way to evaluate the uncertainty in I caused by cross-section uncertainties; i.e.,

$$\left[\frac{\Delta I}{I}\right]^2 = \sum_{all\ xmg} \sum_{all\ x'm'g'} P^g_{xm} P^{g'}_{x'm'} \frac{\mathrm{cov}(\sigma^g_{xm}, \sigma^{g'}_{x'm'})}{\sigma^g_{xm}\,\sigma^{g'}_{x'm'}} \tag{1}$$

In this expression σ^g_{xm} represents the interaction cross section for reactions of type x in material m in energy group g, $\mathrm{cov}(\sigma^g_{xm}, \sigma^{g'}_{x'm'})$ is the covariance matrix for the indicated multigroup cross sections, and P^g_{xm} is the relative sensitivity profile of response I for cross section σ^g_{xm}, as defined by

$$P^g_{xm} = \frac{\partial I/I}{\partial \sigma^g_{xm}/\sigma^g_{xm}} \tag{2}$$

The relative sensitivity profile clearly can be interpreted as the fractional change in the response per fractional cross-section change. Note that in Eq. (1) the first two factors in each term of the summation (the product of sensitivity profile components) is strictly problem dependent, while the third factor involves only cross sections and their covariances and is hence problem independent.

In 1981 the ENDF/B-V 30-group covariance library COVFILS (cf. Ref. [28]) was produced. It contains multigroup cross sections and covariances for individual absorption and scattering reactions, but does not include group-to-group scattering matrices. In 1984 the COVFILS-2 library became available. This major new 74-group library contains not only cross sections and covariances, but also actual P_0 - P_3 scattering matrices for all scattering reactions present in the library, so as to ensure consistency between the library covariances and the scattering matrices used to calculate the corresponding, sensitivity profiles in SENSIBL. The addition of this new matrix data required substantial modifications to SENSIBL, as described below. As a result of both the voluminous covariance data now available and the presence of the large scattering matrices, COVFILS-2 is a rather large file. In order to make its storage and data transfer more manageable, the file was written in a very condensed format called BOXER[27]

that compresses the 7.2 million data elements (which would occupy approximately one million card images in uncondensed form) onto about 40 thousand card images. A set of subroutines called COVARD2 was incorporated into SENSIBL in order to retrieve data in BOXER format. Also, a modification to COVFILS-2 was recently made to expand the special index at the beginning of the file. Details of this modification are discussed below in the section describing the incorporation into SENSIBL of the direct term.

One consequence of the detailed covariance data increasingly becoming available for individual scattering levels is that the SED sensitivity capability is rapidly becoming obsolescent. However, both SED and the corresponding angular distribution sensitivity capabilities are maintained in the new code. Miscellaneous changes since the original SENSIT-2D code include adaptation to CRAY computers (on both the CTSS and COS operating systems), improved efficiency of data transfer, improved architecture, and linkage to the TRISM code. The latter capability now allows performing sensitivity and uncertainty analyses on systems with large void streaming regions, which previously could not be accomplished conveniently with deterministic transport codes. Hence, the sensitivity and uncertainty analysis capability is being kept abreast with the state of the art of both deterministic transport methods and covariance libraries. Additional user-oriented improvements were made to the input and output formats. Most significant of these was the preparation of summary tables in the output listing, an addition made imperative by the sheer mass of covariance data for many materials and cross-section types, as well as the increasingly detailed nucleonic models possible with deterministic transport codes.

We now turn back to the sensitivity profile given by Eq. (2). The sensitivity profile for the response I can be expressed as

$$P^g_{xm} = \frac{1}{I}\left\{ R^g_{xm}\,\varphi^g_m - \sigma^1_{xm}\,\chi^g_m + \sum_{\ell=0}^{LMAX}\sum_{g'=g}^{IGM} \sigma^{g\to g'}_{xm\ell}\,\psi^{gg'}_{m\ell}\right\}, \tag{3}$$

The integral of the response over the volume of the "detector" (which may be the entire system) is

$$I = \sum_{i\ \epsilon\ det} V_i \sum_{all\ g} \phi^{0g}_{0i} \sum_{all\ xm} N_{mi}\,R^g_{xm}, \tag{4}$$

where V_i is the volume of spatial interval i, ϕ^{0g}_{0i} is the scalar flux in group g in interval i, and N_{mi} is the local atomic density of material m in interval i. The quantity R^g_{xm} appearing in Eqs. (3) and (4) is the "response function," a response-weighted microscopic cross section. If the response of interest is the number of nuclear reactions of type x, then R^g_{xm} is just the microscopic cross section for that reaction. However, if the response of interest is the total nuclear heating, for example, then R^g_{xm} is the partial kerma factor (in units of eV-barns) for reaction x in material m in group g. Other complex responses, such as dpa and total helium production, can also be accommodated using suitable definitions of R^g_{xm}.

The quantities φ^g_m, χ^g_m, and $\psi^{gg'}_{m\ell}$ appearing in Eq. (3) are atom density-weighted, spatial integrals of the flux defined in Eqs. (5), (6), and (7), respectively. The quantity φ^g_m is defined as

$$\varphi_m^g = \sum_{i \in det} N_{mi} V_i \phi_{0i}^{0g} . \quad (5)$$

In Equation (3), σ_{xm}^g is the microscopic cross section for reaction x in material m and in group g, $\sigma_{xm\ell}^{g \to g'}$ is the ℓ^{th} Legendre moment of the scattering cross section for energy transfers from group g to group g' for a particular reaction type x in material m. In this same equation, χ_m^g is the numerical integral of the product of forward and adjoint angular fluxes over all angles and all spatial intervals, LMAX is the order of scattering, and MM is the number of angular directions:

$$\chi_m^g = 4\pi \sum_{all\ i} N_{mi} V_i \sum_{n=1}^{MM} \Phi_{ni}^g \Phi_{ni}^{*g} w_n = \sum_{\ell=0}^{LMAX} \psi_{\ell m}^{gg} . \quad (6)$$

Φ_{ni}^g and Φ_{ni}^{*g} are discrete-ordinates representations of forward and adjoint angular fluxes, respectively, for group g, spatial mesh point i and discrete direction n. $\psi_{m\ell}^{gg'}$ is the density-weighted spatial integral of the product of the spherical harmonics expansion for forward and adjoint angular fluxes for material m,

$$\psi_{\ell m}^{gg'} = 4\pi \sum_{all\ i} N_{mi} V_i (2\ell + 1) \sum_{k=0}^{\ell} \phi_{\ell i}^{kg} \phi_{\ell i}^{*kg'} . \quad (7)$$

Direct Term

The first term in the brackets of Eq. (3) is the direct term. Note that this term non-zero only if reaction x in material m contributes to the detector response function, so that both R_{xm}^g and φ_m^g are non-zero.

Incorporating the direct term into the sensitivity profile calculated by SENSIBL is straightforward, because zone-averaged fluxes such as φ_m^g are readily available from other calculations. The code was modified to accept, as input for each detector zone, the information needed to define the detector response function (and source for the adjoint flux calculation); namely, the material number MAT_d, the reaction number MT_d, and the corresponding material density. In most cases the value of MT_d corresponds to the MT number of a single reaction in the COVFILS-2 library. Some complexity is introduced by the need to calculate sensitivities for complex reactions which have direct contributions from several different reactions in the library. An example is the $(n, n't)$ reaction in ^{7}Li, $MT_d = 33$. In terms of reaction MTs explicitly present in the library, this reaction is the sum of reaction MTs 853 through 858. An additional problem in ^{7}Li is that reaction numbers in the 851-870 range are used in ENDF/B to specify evaluator-defined groups or "lumps" of reactions[12]. In this case, the MT number alone does not determine whether or not a given reaction contributes to tritium production, for example. Therefore, the COVFILS-2 index was modified to include a list of important detector reactions (MT_d) to which the library MT makes a direct contribution. Logic was also added to SENSIBL to check, when computing the sensitivity for MAT/MT, whether this MT contributes to MT_d.

If it does, the cross-section vector from COVFILS-2 is used to calculate the direct term. A diagnostic print was also included in SENSIBL to compare the sum of the cross sections found in this manner (for example, MT 853 through 858) to the input value for the detector response function ($MT_d = 33$).

Indirect Term

The second and the third terms of Eq. (3) comprise the indirect term. These terms are called the "loss" term and the "gain" term, respectively. Note that the indirect term receives contributions only from intervals in which the density N_{mi} is non-zero. The indirect term may be derived from the expression for the forward difference approximation, Eq. (36) in Ref. [29] or Eq. (17) in Ref. [30] or Eq. (26) in Ref. [31], considering a two-dimensonal geometry and expansion of the scattering into Legendre polynomial series and the flux angular expansions into the series of spherical harmonics.

The quantities χ^g_m and $\psi^{gg'}_{m\ell}$ in Eqs. (6) and (7) each result from performing a material-density-weighted sum over all spatial intervals. In SENSIT-2D the density term N_{mi} was, in effect, brought outside these sums. This meant that, in a single computer run, sensitivity profiles, and hence uncertainties, could only be calculated for a single zone (domain of constant N_{mi}). To study a complex system having many zones, it was necessary to make multiple runs and then quadratically sum the uncertainties from the different runs. This procedure was time consuming and was only approximately valid, as it omitted the contribution to the total uncertainty from cross-zone correlation terms. In SENSIBL, this restriction has been lifted, and a more accurate region-summed total uncertainty is now calculated in a single computer run. With these modifications made, practical SENSIBL calculations were performed, using input fluxes from TRISM calculations of the LOTUS-facility LBM experiment.

In this process another major gain was made. Not only could we include the cross-zone effect, but the calculational time was also significantly reduced. The 11-zone LBM calculation performed in a single run required only as much time as that required previously for each separate one-zone run.

5 Formulation for the Direct Term for Complex Responses

For covariance analysis[32] of complex responses (such as kerma, dpa or helium production, for example) it is necessary to decompose the complex responses into contributions from individual reactions. These "partial" responses are not directly measurable quantities, but they provide the connection between the total response and the ENDF/B covariances, which are provided for individual reaction cross sections.

We recall from the previous section that σ^g_{xm} is the microscopic cross section (in barns) for reaction x in material m in group g. In the case of complex responses, these reactions will contribute the response l with an effect-weighted cross section R^g_{xm} that differs from σ^g_{xm}. The ratio of the two cross sections we denote by E^g_{xm}, so that

$$R^g_{xm} = \sigma^g_{xm} \cdot E^g_{xm}. \tag{8}$$

E^g_{xm} is thus the effectiveness of these particular nuclear reactions in producing the response I. Later in this section particular examples are discussed which may help clarify these points.

Combining Eqs. (4) and (8), we have

$$I = \sum_{i \ \epsilon \ det} V_i \sum_{all \ g} \phi^{0g}_{0i} \sum_{all \ m} N_{mi} \sum_{all \ x} \sigma^g_{xm} \ E^g_{xm}. \tag{9}$$

The cross sections σ are uncertain, and they influence the integral I in Eq. (9) both directly and through their indirect effect on the fluxes ϕ. The quantities E also are nuclear data, are uncertain, and influence I. However, the current ENDF files do not specify the covariances of charged-particle emission spectra, for example. Thus, for the present, we treat the E-parameters as constants. We return to this point at the end of this section.

Direct Term for Complex Reactions

The direct contribution to the sensitivity profile is, from Eqs. (2) and (9)

$$\begin{aligned} P^g_{xm}(\text{direct}) &= \frac{\sigma^g_{xm}}{I} \frac{\partial I}{\partial \sigma^g_{xm}} \Big|_{direct} \\ &= \frac{\sigma^g_{xm}}{I} \frac{\partial I}{\partial \sigma^g_{xm}} \Big|_{(\phi^{0g}_{0i}, E^g_{xm}) \ \text{constant}}. \end{aligned} \tag{10}$$

¿From Eq. (5) and (9),

$$\frac{\partial I}{\partial \sigma^g_{xm}} \Big|_{(\phi^{0g}_{0i}, E^g_{xm}) \ \text{constant}} = \sum_{i \ \epsilon \ det} V_i \ \phi^{0g}_{0i} \ N_{mi} \ E^g_{xm} = \varphi^g_m \ E^g_{xm}. \tag{11}$$

Combining Eqs. (10) and (11),

$$P^g_{xm}(\text{direct}) = \frac{\sigma^g_{xm}}{I} E^g_{xm} \ \varphi^g_m, \tag{12}$$

or

$$P^g_{xm}(\text{direct}) = \frac{R^g_{xm}}{I} \varphi^g_m. \tag{13}$$

Simple Reaction Rates

In order to illustrate the concepts of "response functions" and "effectiveness," we now consider some specific examples of integral responses. As our first example, we consider a "simple" response, namely, the total number of n,γ events in a specified region. ¿From Eq. (5), we see that φ^g_m will be non-zero only for materials which are present in at least some intervals i of the detector region. Considering only these contributing materials, the effectiveness E^g_{xm} will be unity (for all groups) if x is the reaction index of the n,γ reaction and zero if x is any other reaction. Thus, R^g_{xm} will be equal to σ^g_{xm} or zero, depending on x.

Nuclear Heating

In the case of nuclear heating, the response cross section R^g_{xm} is just the partial kerma due to reaction x in material m in group g. The effectiveness E^g_{xm} is, in this case, the average net charged-particle energy deposited per reaction.

Displacements Per Atom

In the case of dpa, the effectiveness E^g_{xm} is the average number of atoms displaced from their normal lattice positions (due mainly to interactions of the primary recoil nucleus with the lattice) per reaction of type x in material m in group g. Methods for calculating both partial kerma factors and partial dpa-production cross sections are discussed in Ref. [22].

Helium Production

The reason that helium production differs from ordinary reaction rates is that a "multiplicity" is involved. In ENDF/B-V ^{12}C, for example, all inelastic levels above the first one decay via 3α emission. Thus,

$$\sigma(n, x\alpha)\,|_{^{12}C} = \sigma_{107} + 3 \cdot (\sigma_{52} + \ldots + \sigma_{91}) \tag{14}$$

The multiplicity, then, is 1.0 for MT107 and 3.0 for reactions MT52 through MT91. In this case the effectiveness E^g_{xm} is just the multiplicity. The multiplicities for helium production are normally energy-independent integers, although there are exceptions. In ^{7}Li (ENDF/B-V.2) for example, covariances are given for the total $(n, 2n)$ reaction in MT851. This cross section is the sum of the $(n, 2n)$ and $(n, 2n\alpha d)$ reactions. The helium yield per "reaction" here is clearly energy-dependent. As in the case of kerma and dpa, NJOY can provide the separate cross sections for $(n, 2n\alpha d)$ and for MT851, and the group-dependent helium-production multiplicities (E^g_{xm}) can be obtained by division.

It is worth noting that total helium-production cross sections, H^g_m summed over all reactions, are provided directly in ENDF/B-V on Tape 533, along with "integral" covariances, such as

$$\mathrm{cov}(H^g_m, H^{g'}_m) \tag{15}$$

These cross sections and covariances are suitable for thin foil reactor-dosimetry purposes, but they are not very useful for the analysis of fusion-reactor integral experiments, where the dosimetry foil and the transport medium are often made of the same material. When covariance information is presented in "integral" form, as in Eq. (15), the correlation between the individual helium-producing reactions and the reactions important to neutron transport is lost. On the other hand, combining sensitivities and covariances for separate reactions, using Eq. (1), preserves this correlation information.

Mechanics

To perform an uncertainty analysis of a complex reaction, it is clear from Eq. (3) that one needs access to the response cross section R^g_{xm} (or to the E^g_{xm}) for each COVFILS-2 reaction for the materials of interest. Because this information is not present on COVFILS-2, it must be supplied in the user input to SENSIBL. In the case of kerma or dpa, it would be convenient to supply the actual cross sections R^g_{xm}, thereby eliminating the need for a "hand" calculation of E^g_{xm}. For helium production, a mixed strategy is needed. For most reactions in most materials, an energy - independent integer multiplier would be sufficient for constructing R^g_{xm} from σ^g_{xm}, as in the ^{12}C example above. For ^{7}Li, on the other hand, one would like to enter the $(n, 2n\alpha d)$ cross section (R^g_{xm}) from input, just as in the case of kerma. The capability to input a general, energy - dependent multiplier does not seem necessary.

E-parameter Covariances

Our final remarks concern a possible future generalization of this approach. Up to this point, we have considered only cross-section covariances, as are presently contained in COVFILS-2. It may be possible at a later date to add covariances of the effectiveness parameters, E^g_{xm}. If and when this occurs, it will become of interest to calculate the relative sensitivity to these E-parameters, as well as cross sections. Luckily, this will not complicate the coding of SENSIBL very much at all. Examination of Eq. (9) reveals that the direct effect of a fractional change in E^g_{xm} is numerically *equal* to the direct effect of changing σ^g_{xm}.

Since the E-parameters do not effect the neutron flux, there is no indirect term, and the relative sensitivity is obtained immediately from Eq. (13). Only a very few lines of code would be affected in adding an E-parameter sensitivity capability to SENSIBL.

6 Future Plans for SENSIBL

Plans for future work under the Los Alamos/EIR Cooperative Agreement include several tasks related to sensitivity and uncertainty analysis. Of immediate importance is the devising of a simple test case for SENSIBL code verification. Because of the lack of any detailed confirming calculations of uncertainty analyses employing the COVFILS-2 data, there is no experience to date upon which to base "intuitive" judgements on the reasonableness of results. Efforts are now underway to create an exceptionally simple covariance library and a corresponding two-material (^{1}H and ^{6}Li), two-region nucleonic model which would facilitate hand calculations for comparison purposes.

A study is also being conducted of the feasibility of putting multigroup covariance data into future MATXS libraries. One motivation for this study is the requirement to perform sensitivity studies for systems in which temperature dependence and self shielding of the cross sections is required. Both are presently available from the MATXS libraries via the TRANSX-CTR code[33], but not from the COVFILS-2 library. Provision of a MATXS library with

the covariance multigroup data incorporated would permit a more automated approach to the sensitivity and uncertainty analysis of complex responses than is presently possible.

A group collapse routine for the 74-group COVFILS-2 library could be added to TRANSX-CTR at the same time. A collapse capability would allow sensitivity and uncertainty analyses making direct use of transport calculations in, for example, the Los Alamos 30-group structure, which has been employed in some recent analyses of the Japanese FNS blanket experiments.

Another feature expected to be incorporated into SENSIBL in the near future is a one-dimensional option, where the linkage would be to standard flux file output from the ONEDANT code[17]. Thus, one code would serve both one- and two-dimensional requirements.

Also investigated will be the use of CCCC standard interface files[34], such as the ONEDANT and TWODANT files NDXSRF, ZNATDN, and GEODST for nuclide densities, subzone nuclide atomic densities, and geometry description, respectively. TRISM employs a CCCC-like interface file (CCCC standard files are not defined for the TRISM banded, triangular mesh) called GEOMTY for geometry description while CCCC-like versions of NDXSRF and ZNATDN will require further development. These could lead to considerable simplification of SENSIBL input.

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